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September 9, 1997

AEP:NRC:1223E

Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Gentlemen:

Donald C. Cook Nuclear Plant Unit 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING POWER UPRATE AND RELATED CHANGES

This letter and its attachment constitute a response to the July 9, 1997, NRC request for additional information regarding our July 11, 1996, 5% thermal power uprate AEP:NRC:1223 submittal. The request for additional information primarily involves analysis assumptions and methodology.

This letter is submitted pursuant to 10 CFR 50.30(b) and, as such, includes an oath statement.

Sincerely,

E. E. Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 9th DAY OF September, 1997

Linda L. Boelcke
Notary Public

My Commission Expires 1-21-2001

vlb

LINDA L. BOELCKE
Notary Public, Berrien County, MI
My Commission Expires January 21, 2001

Attachment

c: A. A. Blind
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. R. Padgett

A001/1

170040

ATTACHMENT TO AEP:NRC:1223E

Donald C. Cook Nuclear Plant Unit 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING POWER UPRATE AND RELATED CHANGES

NRC QUESTION NO. 1

"In Section 2.0 of Reference 2, you indicated that WCAP-11902 and Supplement were used as the basis for the evaluation of the Unit 2 operation at core power level of 3588 MWt. However, WCAP-11902 licensing report was reviewed and approved by the staff, for D. C. Cook Unit 1 operating at 3250 MWt. Clarify whether the Supplement to WCAP-11902, entitled, "Rerated Power and Revised Temperature and Pressure Operation for Cook Nuclear Plant Units 1 and 2 Licensing Report," was reviewed and approved by the staff for application at the Cook Nuclear Plant (CNP). If not, state the basis of applying these two previous evaluations for all performance parameters between the proposed Unit 2 uprate and the previous rerated program."

RESPONSE TO QUESTION NO. 1

Attachment 5 to AEP:NRC:1223 submittal, from E. E. Fitzpatrick to the USNRC document control desk, dated July 11, 1996, is "Discussion of Previous Related Submissions." The introduction section of attachment 5 addresses, in a general way, the fact that the analyses that support the proposed uprating have been performed over a period of years as a part of other efforts with more immediate short range goals. This attachment states:

- "The analyses that support the proposed uprating of Donald C. Cook Nuclear Plant Unit 2 have been performed over a period of years in several contexts. The analysis of the nuclear steam supply system (NSSS) for an NSSS power of 3600 MWt was performed in conjunction with analyses to operate unit 1 at reduced temperature and pressure (the "Rerating Program"). Most of the core response analyses were performed at an uprated core thermal power of 3588 MWt as a part of the transition from Advanced Nuclear Fuel to Westinghouse Vantage 5 fuel. The recently submitted analyses, AEP:NRC:1207 (erroneously stated to be AEP:NRC:1223 in the submittal), to support an increase in the permitted level of steam generator tube plugging for unit 1 includes a steam mass and energy release analysis to the containment which bounds both units at 3600 MWt. For this submittal (i.e., AEP:NRC:1223), previous NSSS analyses and core response analyses have been reviewed, new analyses have been performed where necessary, and the balance of plant evaluated, as described within this submittal, to support the proposal to increase the core rated thermal power to 3588 MWt."

In particular, as indicated in attachment 5, the supplement to WCAP-11902 was submitted in part in support of a number of proposed technical specification (T/S) changes. It was submitted in its entirety in support of our proposal to reduce the boron concentration in the boron injection tanks of both units to 0 ppm. Our submittal was letter AEP:NRC:1140, "Technical Specification Change Request, Boron Injection Tank (BIT), Boron Concentration Reduction", from M. P. Alexich to T. E. Murley, dated March 26, 1991. The proposal was approved by Amendment No. 158 to Facility Operating License No. DPR-58 and Amendment No. 142 to Facility Operating License No. DPR-74.

NRC QUESTION NO. 2

"Clarify whether the rerating analyses of the pressure transients and the postulated loss-of-coolant accident (LOCA) include the proposed pressurizer safety and relief valve tolerance +/-3%, and the previously NRC-approved main steam safety and relief valves tolerance of +/-3%. If not, state how the rerating analyses applies to the proposed Unit 2 power uprate."

RESPONSE TO QUESTION NO. 2

The analyses performed for submittal AEP:NRC:1223, to increase the thermal power of Cook Nuclear Plant unit 2 to 3588 MWt, assumed setpoint tolerances of 3% for both the pressurizer safety valves and the steam generator safety valves. The pressurizer safety valve setpoint tolerance is specifically addressed for the applicable analyses in section 3.3, "Non-LOCA Analyses", of WCAP-14489, attachment 6 to submittal AEP:NRC:1223. This assumption is called out specifically for the applicable events because this is a new assumption for the unit 2 analyses. The pressurizer pressure setpoint does not affect the LOCA event because the primary system depressurizes. The assumption of a 3% tolerance for steam generator safety valve setpoints was not specifically called out for the new analyses because it is an assumption that was previously submitted and reviewed. An assumption of 3% setpoint tolerance for steam generator safety valve setpoints is input to the applicable analyses in the unit 2 uprate submittal.

NRC QUESTION NO. 3

"Discuss the operability of the safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that the safety-related motor operated valves (MOVs) will be capable of performing their intended functions following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed."

RESPONSE TO QUESTION NO. 3

AFW, CCW, AND ESW SYSTEMS

The safety systems we reviewed for impact from uprated conditions are the auxiliary feedwater (AFW), component cooling water (CCW), and essential service water (ESW) systems. Our review indicates that the mechanical components (i.e., valves and pumps) in these systems are not significantly affected by the uprated power conditions. The performance and T/S requirements for these systems remain unchanged. Because the system parameters have not changed, the associated MOV operability is not impacted.

The following summarizes our review in support of the preceding statement for the indicated systems.

The AFW system provides water to the steam generators when the main feedwater system is unavailable due to a loss of feedwater, unit

trip, feedwater or steam line break, loss of offsite power, or loss-of-coolant accident (LOCA). The AFW system is designed and analyzed to provide sufficient flow to the steam generators during these events against a steam generator pressure corresponding to the set pressure, plus accumulation of the lowest set safety valves. The AFW system is also capable of providing reduced flow at the higher steam generator pressures, plus accumulation corresponding to the higher set safety valves. The uprated conditions did not alter the AFW system's flow requirements or the system's ability to fulfill these requirements. The uprated conditions did not affect or revise the safety valve's set pressure, the AFW pump's operating parameters (flow and head), or the fluid parameters (temperature and pressure). The uprate also did not result in any significant changes in ambient temperatures. Therefore, the AFW's MOV requirements are essentally unchanged, and the mechanical components in the system are not significantly affected.

The CCW system is a closed loop system that serves as an intermediate loop between potentially radioactive systems and lake water to ensure that leakage of radioactive fluid is contained within the plant. The CCW system is designed and analyzed to supply cooling water flow during the injection and recirculation phases of a LOCA and during unit operation. The LOCA long-term mass and energy release and containment integrity analyses performed by Westinghouse utilized CCW system flowrates and heat exchanger UAs representative of the uprated conditions. The Westinghouse analyses determined the results were acceptable for containment integrity pressure and temperature response. These details were provided in our submittal AEP:NRC:1223C, dated June 10, 1997. Based on this, the uprated conditions did not significantly impact the CCW system's heat removal requirements, or the system's capability to meet these requirements. The CCW pumps' operating parameters (flow and head) and fluid parameters (temperature and pressure) were not changed as a result of the uprate. The uprate also did not result in any significant changes in ambient temperatures. Therefore, the CCW's MOV requirements are essentially unchanged and the mechanical components in the system are not significantly affected.

The ESW system provides cooling water requirements to the CCW heat exchangers, emergency diesel generators, CTS heat exchangers, and the control room air conditioning condensers. The ESW system is operated in conjunction with the CCW and CTS systems. The ESW pump's operating parameters (flow and head) and fluid parameters (temperature and pressure) were not changed as a result of the uprate. The uprate also did not result in any significant changes in ambient temperatures. Therefore, the ESW's MOV requirements remain essentially unchanged and the mechanical components in the system are not significantly affected.

RCS, CVCS, AND RHRS SYSTEMS

The safety systems to be reviewed for impact from uprated conditions are the reactor coolant system (RCS), emergency core cooling system (ECCS), and chemical volume control system (CVCS). Our review indicates that the mechanical components in these systems are not significantly affected by the uprated power conditions. The performance and T/S requirements for these systems remain unchanged. Because the system parameters have not changed, the associated MOVs operation is not significantly impacted.

The RCS consists of four identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP) and a steam generator. In addition, the system includes a pressurizer, a pressurizer relief tank, inter-connecting piping, and instrumentation necessary for operational control. During operation, the RCPs circulate pressurized water through the reactor vessel and the four coolant loops. The water, that serves both as a coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the core. It then flows to the steam generators where the heat is transferred to the steam system, and returns to the RCPs to repeat the cycle. The RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Three spring loaded safety valves and three power operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

Fluid systems calculations were performed, evaluating the capability of the RCS to operate at the uprate program conditions. The uprated power conditions did not affect any of the RCS safety related mechanical components design basis. The MOVs fluid system design conditions (fluid flow, temperature, pressure and differential pressure) were not significantly affected by the uprated conditions.

The CVCS provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger, then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant, and the letdown orifice reduces the pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger, followed by a second pressure reduction due to the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

The regenerative and letdown heat exchangers are designed to cool letdown flow from T_{cold} to 115° F. The variations in T_{cold} considered for the uprate program are bounded by the design inlet temperature of 547° F for the regenerative heat exchanger. Therefore, the cooling requirements of the letdown function are met with the revised operating parameters.

The letdown function is designed to reduce the static pressure of the reactor letdown stream from the RCP suction pressure to VCT operating pressure, such that the design pressure of intervening piping and components is not exceeded, and fluid is maintained in a subcooled condition throughout the system. The pressure reduction requirements of the letdown function are met with the revised operating parameters.

The centrifugal charging pump operating conditions have not been impacted by the uprating conditions. Fluid systems calculations were performed evaluating the capability of the CVCS to operate at the uprate program conditions. The uprated power conditions do not

significantly affect the CVCS safety related mechanical components' design bases.

The ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two safety injection (SI) pumps and two residual heat removal pumps take suction from the refueling water storage tank (RWST) and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. At the completion of the injection phase from the RWST the ECCS is then aligned to the containment sump, as the suction source, to provide the cold or hot leg recirculation injection flows.

The primary system pressures considered for this program are less than, or equal to, the primary system pressure against which the original system was designed to deliver. Therefore, the revised primary system parameters do not require an increase in either the motive pressure or core cooling capacity of the ECCS. Fluid systems calculations were performed evaluating the capability of the ECCS to operate at the uprate program conditions. The uprated power conditions did not significantly affect the ECCS safety related mechanical components' design bases.

NRC QUESTION NO. 4

"In reference to Sections 3.11.2 and 3.11.3 of reference 2 (WCAP-14489), provide the maximum calculated stresses and cumulative Usage Factors at the most limiting locations and components of the reactor vessel and internals, steam generator, reactor coolant pump, pressurizer, and control rod drive mechanism. Also provide the allowable code limits, the code, and the code edition used in the evaluation for the power uprate. If different from the code of record, provide the necessary justification."

RESPONSE TO QUESTION NO. 4

Reactor Vessel:

With respect to section 3.11.2, the results of the reactor vessel analyses and evaluations are summarized below. The stress intensity and fatigue usage limits (with the exception of the $3S_m$ maximum range of primary plus secondary stress intensity limit for the control rod drive mechanism (CRDM) housings and outlet nozzle safe end) of the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, with Addenda through the Summer of 1968, are met. The exceeding of the $3S_m$ limit for the CRDM housings and outlet nozzle safe end is reconciled by using the ASME code acceptable method of elastic-plastic analyses in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition.

CRDM Housing

The maximum range of primary plus secondary stress intensity is calculated to be 77.76 ksi, which exceeds the $3S_m$ limit of 69.9 ksi. However, a simplified elastic-plastic analysis was performed in accordance with paragraph NB-3228.3 of the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, and the higher range of stress intensity is reconciled. The maximum cumulative

fatigue usage factor is 0.1687, which is below the ASME code limit of 1.0.

Main Closure Region

The main closure region of the reactor vessel consists of the vessel flange, the closure head flange, and the closure stud assemblies that couple the head to the vessel. The maximum ranges of stress intensity in the closure head flange and the vessel flange are 65.26 ksi and 61.04 ksi, respectively, compared to the ASME code $3S_m$ limit of 80.1 ksi. The maximum service in the closure studs is 91.8 ksi, which compares favorably to the $3S_m$ limit of 107.7 ksi.

The maximum cumulative fatigue usage factor for the closure head flange, vessel flange and closure studs are 0.018, 0.029 and 0.99, respectively. The usage factors are all less than the 1.0 ASME code limit. However, it should be noted that the closure stud usage factor of 0.99 was calculated under the assumption that the first 25% of the 11,680 occurrences of plant loading and unloading, at 5% of full power per minute (2,920 occurrences of each), occurred during the first ten years of operation when the vessel outlet temperature (T_{hot}) was 599.3° F.

Outlet Nozzle

The maximum range of primary plus secondary stress intensity in the outlet nozzle end is calculated to be 59.58 ksi compared to the $3S_m$ limit for austenitic stainless steel material of 50.1 ksi. Because the maximum range of stress intensity exceeds $3S_m$, a simplified elastic-plastic analysis per paragraph NB-3228.3 of the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, was performed that justified the higher maximum range of stress intensity. The maximum usage factor at the safe end is 0.021, which is less than 1.0.

The maximum range of stress intensity in the outlet nozzle and nozzle to shell juncture is 57.09 ksi, compared to the $3S_m$ allowable 80.1 ksi. The maximum cumulative usage factor in the nozzle and nozzle to shell juncture is 0.0631, which is also less than 1.0.

Inlet Nozzle

The maximum range of stress intensity in the inlet nozzle safe end is 49.65 ksi, which is less than $3S_m$ = 50.1 ksi. The maximum range of stress intensity in the inlet nozzle and nozzle to shell juncture is 49.86 ksi, which compares favorably with a $3S_m$ limit of 80.1 ksi. The maximum cumulative usage factors in the nozzle safe end and nozzle to shell juncture are 0.0174 and 0.0977, respectively, which are both less than 1.0.

Vessel Wall Transition

The maximum range of stress intensity and cumulative fatigue usage factor for the vessel wall transition, between the nozzle shell and the vessel beltline, are 33.57 ksi and 0.0066. These values are less than the ASME code limits of 80.1 ksi and 1.0, respectively.

Bottom Head-to-Shell Juncture

The maximum range of primary plus secondary stress intensity at the juncture, between the vessel bottom hemispherical head and the vessel beltline shell, is 34.53 ksi compared to a $3S_m$ allowable of 80.1 ksi. The maximum cumulative fatigue usage factor at the juncture was calculated to be 0.0182, which is less than 1.0.

Bottom Head Instrumentation Penetrations

The bottom head instrumentation penetrations are acceptable for uprating, based upon a maximum range of primary plus secondary stress intensity of 51.49 ksi, and a maximum cumulative usage factor of 0.1220. These values compare favorably with the ASME code allowables of 69.9 ksi ($3S_m$) and 1.0, respectively.

Core Support Pads

The core support pads were evaluated to have a maximum range of stress intensity of 69.7 ksi, compared to a $3S_m$ limit of 69.9 ksi. The maximum cumulative fatigue usage factor was calculated to be 0.693, which is less than the 1.0 ASME code limit.

Reactor Vessel Internals

Cook Nuclear Plant unit 2 reactor internals are composed of two sections, the upper internals and the lower internals. Evaluations were performed for the critical components for both the upper internals and lower internals. The following is a list of the critical components for the upper and lower internals.

Upper Internals

Perforated section of the top hat support structure.

Lower Internals

Lower Support Assembly
Core Barrel and Flange
Lower Radial Support
Clevis Inserts
Baffle-Former Assembly
Upper Core Plate Alignment Pins
Thermal Shield

The structural evaluations performed for the above areas confirmed that their structural integrity and increased fatigue usage was found to be within acceptable limits, according to the original design basis.

Steam Generator:

The unit 2 steam generators were replaced in 1987. The discussion below addresses the replaced components and remaining original upper shell components separately.

Replacement Components

The criteria used to determine acceptable stress states are provided in the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, and the associated Addenda through Winter 1968.

Component	Maximum Stress Calculated	Maximum Stress Allowable	Fatigue Usage Calculated	Fatigue Usage Allowable
Primary Chamber, Tubesheet, Stub Barrel	(1)	(1)	0.13	1.0
Primary Nozzles	31.9 ksi	58.2 ksi	0.87	1.0
Primary Manways	41.0 ksi	48.3 ksi	0.91	1.0
Tubes	47.96 ksi	79.80 ksi	0.59	1.0
Primary Chamber Divider Plate	(1)	(1)	0.19	1.0
Tube to Tubesheet Weld	(1)	(1)	0.75	1.0
Lower Shell/Cone/Upper Shell	79.2 ksi	80.1 ksi	0.12	1.0
Trunnions	58.8 ksi	80.1 ksi	0.01	1.0
Minor Bolted Openings	93.9 ksi	94.3 ksi	0.74	1.0
Minor Nozzles	29.3 ksi	80.1 ksi	0.88	1.0
Internals	(2)	(2)	0.06	1.0
Feedwater Ring and J-Nozzles	26.7 ksi	27.0 ksi	0.56	1.0

(1) The primary + secondary stresses exceed the allowable stress limit of $3S_m$. A plastic analysis was performed per paragraph N-417.6(b) of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels", 1968 Edition with Addenda to and including Winter 1968, code of record, to demonstrate structural integrity.

(2) The maximum stresses in the steam generator internals occur during the faulted conditions. For the normal and upset conditions, the primary + secondary + peak stresses in the steam generator internals are low, and below the endurance limit. Therefore, the maximum fatigue usage for the steam generator internals is 0.06.

Original Upper Shell Components

Primary stresses and maximum stress ranges are not affected by the uprating conditions, and these calculations were not repeated. When considering the upper and lower bound primary temperatures, the upper bound temperature conditions are very close to the transient conditions used in the reference analyses, and the resulting fatigue usages show only slight variations from the reference conditions. However, the lower bound temperature conditions can result in increased fatigue usages in some cases. A summary of the fatigue usages is provided below.

Component	Referenced	Fatigue Usage Upper Bound Temperature	Fatigue Usage Lower Bound Temperature
Main Feedwater Nozzle	0.53	0.724	0.941
Secondary Manway Shell Penetration	0.17	0.051	0.053
Secondary Manway Bolts	(3)	0.427	0.825
Steam Nozzle	0.59	0.616	0.616

(3) The reference value for fatigue is not provided. The stresses used for the analysis of the bolts are taken from another model steam generator, with scale factors to account for geometry variations.

As part of the uprating program, the steam generator structural integrity was evaluated to account for the revised loss of load and loss of offsite power transients. The evaluation showed that the component most affected by the uprating program is the tubesheet-to-channel head junction. The stress intensities continue to satisfy the stress limits. The calculated value of the fatigue usage, 0.34, remains within the maximum allowable limit of 1.0.

Reactor Coolant Pump:

The evaluation performed for the RCPs addressed the ASME code structural considerations for the RCP casing, main flange, main flange bolts, thermal barrier, casing foot, casing discharge, and suction nozzles, casing weir plate, seal housing, and auxiliary nozzles. For unit 2 the ASME Code, Section III, 1968 Edition, with Addenda through Summer 1969, was used as a guide.

The RCP evaluation addressed the revised NSSS parameters and NSSS design transients associated with the uprating, and compared these parameters and transients to the conditions assumed in the original design analyses for the RCPs. The differences (i.e., delta temperatures [DTs] and differential pressures [DPs]) were identified and used to obtain stress and fatigue results for power uprate.

The DPs associated with the power uprate design transients were reviewed to determine if there were any changes that would qualify

as a "significant fluctuation" in accordance with the ASME code definition, and, thus, require consideration relative to fatigue. It was concluded due to the power uprate design transients, that all DPs were less than the ASME code definition of "significant fluctuation" value, and that no fatigue consideration is required because the fatigue waiver remains unchanged. The design transients were then reviewed to identify the maximum pressure to which the RCP could be exposed. For unit 2, this maximum pressure was determined to be 2724.1 psia for the loss of load transient. A review of RCP analyses performed for other plants showed that increases to 2725 psia have been analyzed in detail and shown to be acceptable. It was concluded that the pressure transients are acceptable.

The effect of power uprate on the various original analyses for the RCPs was also assessed using the NSSS design transients and the associated DT values. For the most part, the comparison of NSSS design transients and assessments of associated DT values were sufficient to show continued applicability of the original analyses to power uprate conditions. One area where the increase in DT was sufficient to merit analysis was for the casing weir plate. The evaluation showed a range of stress intensities = 41,379 psi for power uprate conditions. Comparison of this value to the ASME code primary plus secondary stress limit of $3S_m$ = 50,700 psi showed that the ASME code limit is satisfied. Fatigue requirements for the weir plate were satisfied by the fatigue waiver (ASME code, NB-3222.4(d)).

In summary, the results of the power uprate assessments showed that the ASME code criteria are satisfied at power uprate conditions.

Pressurizer:

The external loads are not revised for the 3600 MWt uprating conditions, and the changes in the pressure loads do not affect the previously completed stress calculations. Thus, the primary stresses calculated for the original analysis remain valid at the uprated conditions. Also, the changes in the design transients (loss of load and loss of offsite power) did not have any significant effect on the primary plus secondary stresses. However, for some components, the fatigue analysis is affected. The new calculated fatigue usage factors for each of the pressurizer components are listed below. Because the new calculated fatigue usage factors are less than 1.0, the pressure components meet the stress/fatigue requirements of the ASME Code, Section III, 1965 Edition, including Addenda up to Winter 1966.

PRESSURIZER FATIGUE USAGE FACTORS

Component	Calculated Fatigue Usage
Surge Nozzle	<0.34
Spray Nozzle	0.991
Safety and Relief Nozzle	<0.15
Lower Head, Heater Well	<0.07
Lower Head, Perforation	<0.02
Upper Head and Shell	0.973
Support Skirt/Flange	<0.02
Manway Pad	0.0
Manway Cover	0.0
Manway Bolts	0.0

Support Lug	<0.05
Instrument Nozzle	<0.11
Immersion Heater	<0.01
Valve Support Bracket	0.01

Control Rod Drive Mechanism:

The evaluation performed for the CRDMs addressed the ASME code structural considerations for the pressure boundary components of both the part-length CRDMs, which are not in use, but the pressure boundary components remain present, and the full-length CRDMs. The unit 2 CRDMs were designed and fabricated to the requirements of the 1968 Edition of the ASME Code, Section III. The analysis was based on the criteria contained in the 1971 edition of the ASME Code, Section III. In later editions of Section III (NCA-1140), it is an accepted practice to use a later ASME code edition for analysis of components.

The CRDM evaluation addressed the revised NSSS parameters and NSSS design transients associated with the uprating and compared these parameters and transients to the conditions assumed in the original design analysis for the CRDMs. The differences were identified and used to obtain stress and fatigue results for power uprate. In the original analyses, the component of the pressure housing that experiences the greatest stress range and has the highest fatigue usage is the upper canopy. The DTs and DPs due to uprating were identified and used to establish stress levels using the ratio method based on the original analysis. The thermal and pressure stresses of the original analysis were separated so that the incremental changes from either pressure or temperature could be determined. The results of the evaluation are:

1. The maximum stress intensity range is 109,960 psi, which is less than the maximum allowable range of thermal stress of 127,105 psi determined using the thermal ratchetting requirements of the ASME Code, Section III, NB-3228.

2. The total fatigue usage factor is 0.672, which is less than the usage factor calculated in the original conservative analysis (0.858) and is less than the allowable limit of 1.0 (ASME Code, Section III, 1971 Edition).

In conclusion, based on the numerical evaluation of the stress at the location of the CRDM having the greatest fatigue usage, the CRDM pressure housing meets the requirements of the ASME code at power uprate conditions.

NRC QUESTION NO. 5

"In Table 2.1-1 of Reference 1, the current core power limit is 3391 MWt thermal. On page 2 of Appendix 1 to Reference 1, the group one proposed changes have the current rated core power level of 3411 MWt. Clarify the difference."

RESPONSE TO QUESTION NO. 5

Table 2.1-1 is part of WCAP-14489 that is attachment 6 to our AEP:NRC:1223 submittal. WCAP-14489 was prepared by our contractor, Westinghouse Electric Corporation. The entry indicates the original licensed core power of Cook Nuclear Plant unit 2 was 3391 MWt. This is correct. However, Cook Nuclear Plant's unit 2 was

uprated from a rated thermal power of 3391 MWt to a rated thermal power of 3411 MWt for cycle 4 by Amendment No. 48 to License No. DPR-74. This effort was supported by our contractor, Exxon Nuclear Company, Incorporated. Since Westinghouse did not play a major role in the uprate to 3411 MWt, the authors of WCAP-14489 decided to reference only the original rated thermal power in WCAP-14489.

NRC QUESTION NO. 6

"Discuss the analytical methodology and assumptions used in evaluating pipe supports, nozzles, penetration, guides, valves, pumps, heat exchangers, and support anchors at the uprate conditions. Were the analytical computer codes used in the evaluation different from those used in the original design basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications."

RESPONSE TO QUESTION NO. 6

The uprating program will have an insignificant impact on pipe supports, guides, and anchors. That is, the resultant primary and secondary side temperatures are only slightly higher than the original design basis temperatures. This small temperature rise will result in minimal increases in the forces that the supports, guides, and anchors will experience. These increases are well within the substantial design margins for the components. Thus, the slight increase in temperature will not result in a deviation from the original design bases of the supports, guides, and anchors. No new computer codes were used for this review.

As detailed in our response to question no. 3, the safety systems reviewed for impact from the uprate conditions were the AFW, CCW, and ESW systems. This review indicated that the pumps and valves are not significantly affected by the uprated power conditions because the original design basis performance and T/S requirements remain unchanged. The ESW and CCW systems were analyzed, utilizing the Proto-Flo computer code in order to determine the system inputs used by Westinghouse. The use of the system inputs was detailed in our AEP:NRC:1223C submittal, dated June 10, 1997. Details of the Proto-Flo computer code were discussed in our AEP:NRC:1238F1 submittal, dated April 10, 1997, which was our reply to a request for additional information on calculations provided to the NRC during a SOPI inspection.

The Westinghouse systems evaluated are the: 1) reactor coolant system (RCS); 2) chemical and volume control system (CVCS); 3) emergency core cooling system (ECCS); and 4) residual heat removal system (RHRS).

The fluid systems computer codes used in this evaluation were the:

1. RHRCOOL Code used to evaluate the RHRS cooldown capabilities, and

2. TSHXB heat exchanger code used to evaluate the heat exchanger performance.

The analytical methodology in the computer codes is not different than the original design basis code. These computer codes are in

the Westinghouse quality program described in the energy systems business unit policy and procedures.

Spent Fuel Pool Decay Heat Analysis Method

All spent fuel pool decay heat calculations were performed using implementations of the ORIGEN2 computer code developed at Oak Ridge National Laboratory. This program has a long history of use in the commercial nuclear power industry for both isotope production and thermal power calculations. The ORIGEN2 code is a rigorous isotope generation and depletion code that accurately predicts the products and by-products of fission and the resulting heat generation rates.

The decay heat generation rate in the pool consists of two components: the decay heat generated by previously discharged fuel assemblies, and the decay heat generated by freshly (recently) discharged assemblies. The decay heat contribution of previously discharged fuel assemblies changes very little over short periods of time, and is, therefore, held constant in the analyses. Because of the nature of exponential decay, this simplification is conservative. The Holtec QA Validated LONGOR computer program, which incorporates the ORIGEN2 code, was used to calculate this decay heat component.

The decay heat contribution of the freshly discharged fuel assemblies changes substantially over even very short periods of time. This decay heat contribution is therefore evaluated as time-varying. The Holtec QA Validated BULKTEM computer program, that incorporates the ORIGEN2 code, was used to calculate this decay heat component.

Bulk Spent Fuel Pit (SFP) Temperature Analysis Method

Due to the time-varying decay heat component, the total decay heat is also time-varying. The bulk SFP temperature is therefore calculated as a function of time. The following energy balance is solved to obtain the temperature at each instant in time:

$$C \times \frac{\partial T}{\partial \tau} = Q_{GEN}(\tau) - Q_{HX}(T) - Q_{EVAP}(T)$$

where:

C is the SFP thermal capacity, Btu/°F
T is the bulk SFP temperature, °F
τ is the time after reactor shutdown, hr
$Q_{GEN}(\tau)$ is the decay heat generation, Btu/hr
$Q_{HX}(T)$ is the SFPCS heat rejection, Btu/hr
$Q_{EVAP}(T)$ is the evaporative heat loss, Btu/hr

The evaporative heat loss term includes both evaporative and sensible heat transfer from the surface of the SFP. The implementation of this term has been benchmarked against actual in-plant test data. The solution of this first-order ordinary differential equation is performed using the BULKTEM program.

Time-to-Boil Analysis Method

Following a loss of forced cooling, the continuing decay heat load in the SFP will cause the bulk SFP temperature to rise. The equation energy balance that defines this transient phenomena is

similar to the ordinary differential equation presented above, but does not include the Q_{HX} term and does include a time-varying SFP thermal capacity, to account for the evaporative water losses. The time available for corrective action before bulk SFP boiling occurs is determined using the Holtec QA Validated TBOIL computer program.

The decay heat generation and evaporative heat loss terms in this formulation are identical to those defined above, except for the following two differences:

> The decay heat is calculated using the correlations of USNRC Branch Technical Position ASB 9-2 instead of ORIGEN2.
>
> No incremental credit is given for evaporative heat loss at SFP bulk temperatures greater than 170° F.

Local Temperatures Analysis Method

The decay heat generated by the fuel assemblies stored in the SFP induced a buoyancy driven flow field upward through the fuel rack cells. Cooler water is supplied to the bottom of the racks cells through the rack-to-wall gaps and rack-to-floor plenum. The Holtec QA Validated THERPOOL computer program was used to perform this analysis.

NRC QUESTION NO. 7

"Discuss the effect of flow induced vibration on the steam generator U-bend tubes and the heat exchanger in consideration of high flow rate required for the power uprate."

RESPONSE TO QUESTION NO. 7

The steam generators evaluated for Cook Nuclear Plant's unit 2 uprating program are the replacement model 51F series. A complete U-bend fatigue evaluation was not necessary because of the advanced design features incorporated into the replacement steam generators. One of the prerequisites for excessive U-bend tube fatigue is denting in the top tube support plate. The quatrefoil stainless steel design is expected to inhibit future denting. In addition, the anti-vibration bars (AVBs) incorporated into the replacement steam generators were inserted to a uniform depth three rows deeper than conventional steam generators. Uniform insertion inhibits local flow peaking, and deeper insertion adds margin to calculated tube stability ratios for the largest radius tube not supported by AVBs. Both these factors reduce the risk of fluid elastic tube vibration, which could lead to excessive U-bend tube fatigue.

Flow induced tube vibration and wear analysis for Cook Nuclear Plant's unit 2 model 51F replacement steam generators references normal design loads for operation at 852.75 MWt per steam generator plus consideration of a range of operating conditions for which operation is approved at 900 MWt per steam generator. The main impact of the range of operating conditions was the range of operating pressures considered, so explicit calculations primarily address pressure loading effects that add to the 852.75 MWt base. Calculated results for the advanced model 51F design yield large margins relative to fluid elastic instability limits: the maximum stability ratio is 0.36 versus a limit of 1.00. Uprating from 852.75 to 900 MWt would increase the limiting stability ratio to only 0.38; a result that is still more than 2.5 times below the

limit. Corresponding displacements due to turbulence in the flow are well below 0.001 inch.

Based on these considerations, the replacement steam generators at Cook Nuclear Plant's unit 2 are considered to be effectively designed for the high flow rates required for the power uprate to 3600 MWt.